[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[NRC-2017-0238]

Biweekly Notice

Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations

AGENCY: Nuclear Regulatory Commission.

ACTION: Biweekly notice.

SUMMARY: Pursuant to Section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (NRC) is publishing this regular biweekly notice. The Act requires the Commission to publish notice of any amendments issued, or proposed to be issued, and grants the Commission the authority to issue and make immediately effective any amendment to an operating license or combined license, as applicable, upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from December 5, 2017, to December 18, 2017. The last biweekly notice was published on December 19, 2017.

DATES: Comments must be filed by **[INSERT DATE 30 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. A request for a hearing must be filed by **[INSERT DATE 60 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**.

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2017-0238. Address questions about NRC dockets to Carol Gallagher; telephone: 301-415-3463; e-mail: Carol.Gallagher@nrc.gov. For technical questions, contact the individual listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **Mail comments to:** May Ma, Office of Administration, Mail Stop: OWFN-2-A13, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Lynn Ronewicz, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; telephone: 301-415-1927, e-mail: Lynn.Ronewicz@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID NRC-2017-0238, facility name, unit number(s), plant docket number, application date, and subject when contacting the NRC about the availability of information for this action. You may obtain publicly-available information related to this action by any of the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID NRC-2017-0238.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in this document

- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments

Please include Docket ID NRC-2017-0238, facility name, unit number(s), plant docket number, application date, and subject in your comment submission.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at http://www.regulations.gov as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment into ADAMS.

II. Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Combined Licenses and Proposed No Significant Hazards Consideration Determination.

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in § 50.92 of title 10 of the *Code of Federal Regulations* (10 CFR), this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period if circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. If the Commission takes action prior to the expiration of either the comment period or the notice period, it will publish in the *Federal Register* a notice of issuance. If the Commission makes a final no significant hazards consideration determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

A. Opportunity to Request a Hearing and Petition for Leave to Intervene.

Within 60 days after the date of publication of this notice, any persons (petitioner) whose interest may be affected by this action may file a request for a hearing and petition for leave to intervene (petition) with respect to the action. Petitions shall be filed in accordance with the Commission's "Agency Rules of Practice and Procedure" in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.309. The

NRC's regulations are accessible electronically from the NRC Library on the NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/cfr/. Alternatively, a copy of the regulations is available at the NRC's Public Document Room, located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. If a petition is filed, the Commission or a presiding officer will rule on the petition and, if appropriate, a notice of a hearing will be issued.

As required by 10 CFR 2.309(d) the petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements for standing: (1) the name, address, and telephone number of the petitioner; (2) the nature of the petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the petitioner's interest.

In accordance with 10 CFR 2.309(f), the petition must also set forth the specific contentions which the petitioner seeks to have litigated in the proceeding. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner must provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to the specific sources and documents on which the petitioner intends to rely to support its position on the issue. The petition must include sufficient information to show that a genuine dispute exists with the applicant or licensee on a material issue of law or fact. Contentions must be limited to matters within the scope of the proceeding. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to satisfy the requirements at 10 CFR 2.309(f) with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene. Parties have the opportunity to participate fully in the conduct of the hearing with respect to resolution of that party's admitted contentions, including the opportunity to present evidence, consistent with the NRC's regulations, policies, and procedures.

Petitions must be filed no later than 60 days from the date of publication of this notice. Petitions and motions for leave to file new or amended contentions that are filed after the deadline will not be entertained absent a determination by the presiding officer that the filing demonstrates good cause by satisfying the three factors in 10 CFR 2.309(c)(1)(i) through (iii). The petition must be filed in accordance with the filing instructions in the "Electronic Submissions (E-Filing)" section of this document.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to establish when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, then any hearing held would take place before the issuance of the amendment unless the Commission finds an imminent danger to the health or safety of the public, in which case it will issue an appropriate order or rule under 10 CFR part 2.

A State, local governmental body, Federally-recognized Indian Tribe, or agency thereof, may submit a petition to the Commission to participate as a party under 10 CFR 2.309(h)(1). The petition should state the nature and extent of the petitioner’s interest in the proceeding. The petition should be submitted to the Commission no later than 60 days from the date of publication of this notice. The petition must be filed in accordance with the filing instructions in the “Electronic Submissions (E-Filing)” section of this document, and should meet the requirements for petitions set forth in this section, except that under 10 CFR 2.309(h)(2) a State, local governmental body, or federally recognized Indian Tribe, or agency thereof does not need to address the standing requirements in 10 CFR 2.309(d) if the facility is located within its boundaries. Alternatively, a State, local governmental body, Federally-recognized Indian Tribe, or agency thereof may participate as a non-party under 10 CFR 2.315(c).

If a hearing is granted, any person who is not a party to the proceeding and is not affiliated with or represented by a party may, at the discretion of the presiding officer, be permitted to make a limited appearance pursuant to the provisions of 10 CFR 2.315(a). A person making a limited appearance may make an oral or written statement of his or her position on the issues but may not otherwise participate in the proceeding. A limited appearance may be made at any session of the hearing or at any prehearing conference, subject to the limits and conditions as may be imposed by the presiding officer. Details regarding the opportunity to make a limited appearance will be provided by the presiding officer if such sessions are scheduled.

B. Electronic Submissions (E-Filing).

All documents filed in NRC adjudicatory proceedings, including a request for hearing and petition for leave to intervene (petition), any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities that request to participate under 10 CFR 2.315(c), must be filed in accordance with the NRC's E-Filing rule (72 FR 49139; August 28, 2007, as amended at 77 FR 46562, August 3, 2012). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Detailed guidance on making electronic submissions may be found in the Guidance for Electronic Submissions to the NRC and on the NRC Web site at *http://www.nrc.gov/site-help/e-submittals.html*. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at *hearing.docket@nrc.gov*, or by telephone at 301-415-1677, to (1) request a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign submissions and access the E-Filing system for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a petition or other adjudicatory document (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals/getting-started.html*. Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit adjudicatory documents. Submissions must be in Portable Document Format (PDF). Additional guidance on PDF submissions is available on the NRC's public Web site at *http://www.nrc.gov/site-help/electronic-sub-ref-mat.html*. A filing is considered complete at the time the document is submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC's Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the document on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before adjudicatory documents are filed so that they can obtain access to the documents via the E-Filing system.

A person filing electronically using the NRC's adjudicatory E-Filing system may seek assistance by contacting the NRC's Electronic Filing Help Desk through the "Contact Us" link located on the NRC's public Web site at *http://www.nrc.gov/site-help/e-submittals.html*, by e-mail to *MSHD.Resource@nrc.gov*, or by a toll-free call at 1-866-672-7640. The NRC Electronic Filing Help Desk is available between 9 a.m. and 6 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing stating why there is good cause for not filing electronically and requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing adjudicatory documents in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at *https://adams.nrc.gov/ehd*, unless excluded pursuant to an order of the Commission or the presiding officer. If you do not have an NRC-issued digital ID certificate as described above, click cancel when the link requests certificates and you will be automatically directed to the NRC's electronic hearing dockets where you will be able to access any publicly available documents in a particular hearing docket. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or personal phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. For example, in some instances, individuals provide home addresses in order to demonstrate proximity to a facility or site. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

For further details with respect to these license amendment applications, see the application for amendment which is available for public inspection in ADAMS and at the NRC's PDR. For additional direction on accessing information related to this document, see the "Obtaining Information and Submitting Comments" section of this document.

Dominion Nuclear Connecticut, Inc. (DNC), Docket No. 50-336, Millstone Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: October 4, 2017. A publicly-available version is in ADAMS under Accession No. ML17284A179.

Description of amendment request: The amendment would revise the Millstone Power Station, Unit No. 2 (MPS2) Technical Specification (TS) 6.19, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," with a reference to Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and the limitations and conditions specified in NEI 94-01, Revision 2-A, as the implementing documents used to develop the MPS2 performance-based leakage testing program in accordance with 10 CFR, Appendix J, Option B, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The amendment would allow DNC to extend the Type A primary containment integrated leak rate test interval (ILRT) for MPS2 to 15 years and the Type C local leak rate test interval to 75 months, and incorporates the regulatory positions stated in RG 1.163.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment involves changes to the MPS2 Containment Leakage Rate Testing Program. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators.

   Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, and the limitations and conditions specified in NEI 94-01, Rev. 2-A, for development of the MPS2 performance-based leakage testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less than the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem [roentgen equivalent man] per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in RG 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. DNC has determined that the increase in Conditional Containment Failure Probability due to the proposed change is very small.

Therefore, [the proposed change does not involve a significant increase in the probability or consequences] of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, and the limitations and conditions specified in NEI 94-01, Rev. 2-A, for development of the MPS2 performance-based leakage testing program, and establishes a 15-year interval for Type A testing and an interval not to exceed 75 months for Type C testing. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident; do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, and the limitations and conditions specified in NEI 94-01, Rev. 2-A, for the development of the MPS2 performance-based leakage testing program, and establishes a 15-year interval for Type A testing and an interval not to exceed 75 months for Type C testing. This amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific

> requirements and conditions of the Containment Leakage Rate Testing Program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests will be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A, and the limitations and conditions specified in NEI 94-01, Rev. 2-A.
>
> Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current MPS2 PRA [probabilistic risk assessment] model concluded that extending the ILRT test interval from 10 years to 15 years results in a small change to the MPS2 risk profile.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lillian M. Cuoco, Senior Counsel, Dominion Energy, Inc., 120 Tredegar Street, RS-2, Richmond, VA 23219.

NRC Branch Chief: James G. Danna.

DTE Electric Company, Docket No. 50-341, Fermi 2, Monroe County, Michigan

Date of amendment request: August 24, 2017. A publicly-available version is in ADAMS under Accession No. ML17237A176.

Description of amendment request: The proposed amendment revises Technical Specification (TS) 3.3.1.1, “Reactor Protection System (RPS) Instrumentation,” to eliminate the main steam line radiation monitor (MSLRM) functions for initiating (1) a reactor protection system automatic reactor trip and (2) the associated (Group 1) primary containment isolation system (PCIS) isolations, which include automatic closure of the main steam isolation valves (MSIV) and main steam line (MSL) drain valves. The proposed changes also remove requirements for Group 1 PCIS isolation from TS 3.3.6.1, “Primary Containment Isolation Instrumentation.” This submittal also proposes the addition of two new TS Limiting Conditions for Operation, 3.3.7.2 and 3.3.7.3, for the mechanical vacuum pump and gland seal exhauster trip instrumentation that will be required to actuate in response to high MSL radiation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes eliminate the MSLRM trip and isolation functions from initiating an automatic reactor scram and automatic closure of the MSIVs. The justification for eliminating the MSLRM trip and MSIV isolation functions is based on the NRC-approved evaluation provided in GE LTR [General Electric Licensing Topical Report] NEDO-31400A, “Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor,” dated October 1992.

   The MSLRM high radiation RPS scram function has never been credited to shut down the reactor in response to a postulated CRDA [control rod drop accident]; instead, the neutron monitoring system will continue to be the credited means to shut down the reactor in response to the high flux condition that results from the reactivity inserted by the CRDA.

The consequences of an accident previously evaluated, have been re-evaluated consistent with RG [Regulatory Guide] 1.183 Rev. 0 AST [alternate source term] (10 CFR 50.67) for the applicable DBA [design basis accident] (i.e., the CRDA) as stipulated in NEDO-31400A. The supporting dose analyses demonstrate that, with continued credit for the automatic trip/isolation of the MVPs [mechanical vacuum pump] as well as a new proposed automatic trip of the GSEs [gland seal exhauster], the consequences of the accident are within the regulatory acceptance criteria recommended in RG 1.183 Rev. 0 for compliance with 10 CFR 50.67. As a result, the consequences of any accident previously evaluated are not significantly increased.

The proposed modification of the trip logic for the MVPs to utilize the safety-related MSLRM signals is an improvement over the current licensed configuration of the MVP trip, which utilizes the nonsafety-related offgas 2-minute delay pipe radiation monitor “High-High” radiation signal. Reliance on the safety-related MSLRM signal is consistent with similar approved license amendments and, in addition to improving the quality and reliability of the sensing circuit, ensures the signal is generated at the time of earliest possible detection and therefore improves the effectiveness of the actuation. The trip setpoint utilized corresponds to the same value previously assigned for initiating MSIV isolation in response to the design basis CRDA. The offgas 2-minute delay pipe radiation monitor alarm function is being retained, with a more conservative setpoint, to continue to provide indication of increased radiation.

Similar to the MVPs, the proposed new trip of the nonsafety-related GSEs is also necessary to ensure calculated radiological consequences remain within the regulatory acceptance limits. Reliance on the safety-related MSLRM signal is consistent with BWR [boiling water reactor] design for reliable tripping of the nonsafety-related MVPs and ensures the signal is reliably generated at the time of earliest possible detection and maximizes the effectiveness of the actuation.

The proposed changes also include the elimination of the MSLRM isolation function from automatically closing the MSL drain valves. The contents of the MSL drain lines are conveyed to the main condenser. The evaluation of the condenser release path assumes that 100% of CRDA activity released is transported to the main condenser in 1 second, and therefore, the transportation of the post-CRDA activity from the reactor coolant to the main condenser either via MSLs or MSL drain lines is inconsequential and is supported by the dose analyses performed in support of this submittal.

Neither the MSLRMs nor the MVPs are postulated initiators of any accident previously evaluated. None of the proposed changes alter the probability of the occurrence of the CRDA initiating event.

The loss of the GSEs is a malfunction of equipment considered in UFSAR [updated final safety analysis report] Section 15.12 “Malfunction of Turbine Gland Sealing System.” In the event that the operating blower malfunctions, the backup blower will automatically assume the gas removal requirements. Assuming loss of both blowers, vacuum will be lost in the gland steam condenser. No cladding perforations result from a malfunction of the turbine gland sealing system. The pressure in the gland steam exhaust header will increase to greater than atmospheric, allowing sealing steam to escape into the turbine building. If exhauster vacuum falls below a specified value, caused for example by loss of alternating current (AC) power, a vacuum switch initiates the closing of the live steam supply to the gland steam header. Above 50% to 60% reactor power, the turbine is self-sealing; hence, the packing lines would remain pressurized under normal operating conditions.

The logic associated with the new trip of the GSEs will be designed to preserve the existing ability of the backup exhauster to automatically respond to a loss of the operating exhauster, in the absence of a valid high MSL radiation trip signal. Similar to the design of the RPS trip logic that is proposed to be eliminated, the GSE trip logic will be configured such that no single failure of a MSLRM can generate a GSE trip signal. As specified in the “Applicability” section for the new proposed LCO [limiting condition for operation] 3.3.7.3, the trip logic will be automatically bypassed when reactor power is above 10% RTP [rated thermal power] when the consequences postulated in association with a CRDA are not credible. On the basis of the configuration of the GSE trip logic, the quality of the initiating trip logic signal, and the short duration of normal operation for which the GSE trip logic will be active, the probability of a malfunction of equipment leading to the loss of the turbine gland sealing system is not significantly increased.

The proposed changes do not increase system or component pressures, temperatures, or flowrates for systems designed to prevent accidents or mitigate the consequences of an accident. Since these conditions do not change, the probability of a process-induced failure or malfunction of a SSC [system, structure, or component] is not increased.

The addition of MVP and GSE SRs [surveillance requirements] and LCOs to the TS enhances the reliability of these design functions by establishing administrative requirements for periodic verification of their operability.

The reliance on a lower assigned MSL high radiation alarm setpoint of 1.5 times the full power N-16 background will direct the control room operators to diagnose and act to mitigate conditions associated with fuel damage and release sooner than the current alarm condition which will reduce the potential consequences of a postulated release due to a CRDA.

On the basis of the above considerations, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not increase system or component pressures, temperatures, or flowrates. Since these conditions do not change, the likelihood of a process-induced failure or malfunction of a SSC not previously considered is not increased.

The reliance on the MVP trip to ensure acceptable dose consequences following a postulated CRDA is consistent with the original plant design and licensing bases. The re-assignment of the initiating input for the MVP trip logic to the MSLRM improves the quality and reliability of the credited trip initiating logic by relying on safety-related, redundant components. The quality of the nonsafety-related trip circuit itself is unchanged.

The reliance on the proposed trip of the GSEs is a function that is credited to ensure acceptable dose consequences following a postulated CRDA. The use of the safety-related redundant MSLRM signals and nonsafety-related trip circuit provides the same level of quality and reliability of the initiating trip logic and trip circuitry credited to trip the MVPs. These requirements provide the reliability necessary to ensure the assumptions of the analyzed CRDA remain valid.

Both the safety-related trip logic and the nonsafety-related trip circuits associated with the MVP and GSE trips will be designed to include qualified electrical isolation necessary to ensure the nonsafety-related trip circuitry cannot induce failures of or affect the reliability of the safety-related trip logic.

The new GSE trip will be designed to preserve the existing function for auto-start of the standby exhauster in the event that the plant experiences a loss of the operating exhauster, in the absence of a valid high MSL radiation trip signal. An installed automatic bypass of the GSE trip is actuated once steam flow and feedwater flow correspond to the same Low Power Setpoint used to disable the rod block function of the Rod Worth Minimizer during plant startup. This bypass will minimize the potential for the plant to experience a loss of both GSEs and potential ensuing turbine trip due to a failure of the new trip circuit. The status of the GSE trip bypass will be available to the control room operators and be required to be verified as a part of the plant general operating procedures for startup/shutdown.

Adding requirements for the MVP and GSE trip instrumentation in the TS will ensure that appropriate measures and requirements are in place such that any release of radioactive material released from a gross fuel failure will be contained in the main condenser and processed through the offgas system in the manner credited in the plant analysis of the CRDA.

The MSLRM trip and isolation functions being eliminated as described above are only applicable to the CRDA and no other event in the safety analysis. The proposed changes are consistent with the revised safety analysis assumptions for a CRDA as described in this license amendment request.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes eliminating the MSLRM trip and isolation functions from initiating an automatic reactor scram and automatic closure of the MSIVs are justified based on the NRC-approved LTR NEDO-31400A and supporting dose analysis. The supporting dose analysis also supports the elimination of the MSL drain isolation function of the MSLRMs on the basis that with the valves open the source term associated with the analyzed release is directed to the main condenser the same as it would be via the MSLs themselves.

> The methods of analysis and assumptions used to evaluate the consequences of the applicable impacted safety analysis (i.e. the CRDA) are consistent with the conservative regulatory requirements and guidance identified in Section 5.1 above [this is a reference to “Applicable Regulatory Requirements / Criteria” in DTE August 24, 2017, license amendment request] and establish estimates of the EAB [exclusion area boundary], LPZ [low population zone], and MCR [main control room] doses that comply with these criteria. Hence, there is reasonable assurance that Fermi 2, modified as proposed by this submittal, will continue to provide sufficient safety margins to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters.
>
> Adding requirements for the MVP and GSE high MSL radiation trips in the Fermi 2 TS will ensure that appropriate measures and requirements are in place to maintain the operability of these functions as such that any release of radioactive material from a gross fuel failure resulting from a CRDA will be contained in the main condenser and processed through the offgas system.
>
> The proposed changes do not increase system or component pressures, temperatures, or flowrates for systems designed to prevent accidents or mitigate the consequences of an accident.
>
> The analyses performed in accordance with the specified NRC-approved methods and assumptions demonstrate that the removal of the trip and isolation functions as described will not cause a significant reduction in the margin of safety, as the resulting offsite dose consequences are being maintained within regulatory limits. The proposed changes do not exceed or alter a design basis or a safety limit for a parameter to be described or established in the UFSAR [updated final safety analysis report].
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Jon P. Christinidis, DTE Energy, Expert Attorney - Regulatory, 688 WCB,

One Energy Plaza, Detroit, MI 48226-1279.

NRC Branch Chief: David J. Wrona.

Duke Energy Progress, LLC (Duke Energy), Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Wake and Chatham Counties, North Carolina

Duke Energy Progress, LLC, Docket No. 50-261, H. B. Robinson Steam Electric Plant Unit No. 2 (RNP), Darlington County, South Carolina

Date of amendment request: October 19, 2017. A publicly-available version is in ADAMS under Accession No. ML17292A040.

Description of amendment request: The proposed amendment request consists of five changes that would revise the Technical Specifications (TSs) to support the allowance of Duke Energy to self-perform core reload design and safety analyses. These changes would (1) add the NRC-approved COPERNIC Topical Report (TR) to the list of TRs for HNP and RNP; (2) relocate several TS parameters to the Core Operating Limits Reports for HNP and RNP; (3) revise the RNP TS Moderator Temperature Coefficient maximum upper limit; (4) revise the HNP TS definition of Shutdown Margin consistent with Technical Specifications Task Force (TSTF) Traveler TSTF-248, Revision 0, “Revise Shutdown Margin Definition for Stuck Rod Exception” (ADAMS Accession No. ML040611010); and (5) revise the RNP and HNP power distribution limits limiting condition for operation actions and surveillance requirements to allow operation of a reactor core designed using the DPC-NE-2011-P [proprietary], “Nuclear Design

Methodology Report for Core Operating Limits of Westinghouse Reactors," methodology. (A redacted version, designated as DPC-NE-2011, is publicly-available under ADAMS Accession No. ML16125A420.)

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   COPERNIC
   The proposed change adds a topical report for an NRC-reviewed and approved fuel performance code to the list of topical reports in RNP and HNP Technical Specifications (TS), which is administrative in nature and has no impact on a plant configuration or system performance relied upon to mitigate the consequences of an accident. The list of topical reports in the TS used to develop the core operating limits does not impact either the initiation of an accident or the mitigation of its consequences.

   Relocate TS Parameters to the COLR
   The proposed change relocates certain cycle-specific core operating limits from the RNP and HNP TS to the Core Operating Limits Report (COLR). The cycle-specific values must be calculated using the NRC approved methodologies listed in the COLR section of the TS. Because the parameter limits are determined using the NRC methodologies, they will continue to be within the limit assumed in the accident analysis. As a result, neither the probability nor the consequences of any accident previously evaluated will be affected.

   RNP MTC TS Change
   The proposed change revises the RNP Technical Specification maximum upper Moderator Temperature Coefficient (MTC) limit. Revision of the MTC limit does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

HNP TSTF-248

The proposed change revises the HNP Technical Specification definition of Shutdown Margin (SDM) consistent with existing NRC-approved definition. The proposed revision to the SDM definition will result in analytical flexibility for determining SDM. Revision of the SDM definition does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

DPC-NE-2011-P TS Changes

The proposed change revises the RNP and HNP TS to allow operation of a reactor core designed using the DPC-NE-2011-P methodology. The DPC-NE-2011-P methodology has already been approved by the NRC for use at RNP and HNP. Revision of the TS to align with the NRC-approved methodology does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

COPERNIC

The proposed change adds a topical report for an NRC-reviewed and approved fuel performance code to the list of topical reports in HNP and RNP TS, which is administrative in nature and has no impact on a plant configuration or on system performance. The proposed change updates the list of NRC-approved topical reports used to develop the core operating limits. There is no change to the parameters within which the plant is normally operated. The possibility of a new or different kind of accident is not created.

Relocate TS Parameters to the COLR

The proposed change relocates certain cycle-specific core operating limits from the RNP and HNP TS to the COLR. No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analyses. The proposed changes are consistent with the safety analyses assumptions and current plant operating practice.

RNP MTC TS Change

The proposed change revises the RNP Technical Specification maximum upper MTC limit. The proposed change does not physically alter the plant; that is, no new or different type of equipment will be installed. Therefore the proposed change could also not initiate an equipment malfunction that would result in a new or different type of accident from any previously evaluated. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

HNP TSTF-248

Revising the HNP Technical Specification definition of SDM would not require revision to any SDM boron calculations. Rather, it would afford the analytical flexibility for determining SDM for a particular circumstance. The proposed change does not physically alter the plant; that is, no new or different type of equipment will be installed. Therefore the proposed change could also not initiate an equipment malfunction that would result in a new or different type of accident from any previously evaluated. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

DPC-NE-2011-P TS Changes

The proposed change revises the RNP and HNP TS to allow operation of a reactor core designed using the DPC-NE-2011-P methodology. The DPC-NE-2011-P methodology has already been approved by the NRC for use at RNP and HNP. The proposed change does not physically alter the plant, that is, no new or different type of equipment will be installed. Therefore the proposed change could also not initiate an equipment malfunction that would result in a new or different type of accident from any previously evaluated. Operating the reactor in accordance with the NRC-approved methodology will ensure that the core will operate within safe limits. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment system.

<u>COPERNIC</u>
The proposed change adds a topical report for an NRC-reviewed and approved fuel performance code to the list of topical reports in HNP and RNP TS, which is administrative in nature and does not amend the cycle specific parameters presently required by the TS. The individual TS continue to require operation of the plant within the bounds of the limits specified in the COLR. The proposed change to the list of analytical methods referenced in the COLR does not impact the margin of safety.

<u>Relocate TS Parameters to the COLR</u>
The proposed change relocates certain cycle-specific core operating limits from the RNP and HNP TS to the COLR. This change will have no effect on the margin of safety. The relocated cycle-specific parameters will continue to be calculated using NRC-approved methodologies and will provide the same margin of safety as the values currently located in the TS.

<u>RNP MTC TS Change</u>
The proposed change revises the RNP Technical Specification maximum upper MTC limit. The MTC limit change does not impact the reliability of the fission product barriers to function. Radiological dose to plant operators or to the public will not be impacted as a result of the proposed change. The current Updated Final Safety Analysis Report (UFSAR) Chapter 15 analyses of record remain bounding with the proposed change to the maximum upper MTC limit. Therefore, all of the applicable acceptance criteria continue to be met for each of the analyses with the revised maximum upper MTC limit.

<u>HNP TSTF-248</u>

> The proposed revision to the HNP Technical Specification definition of SDM does not impact the reliability of the fission product barriers to function. Radiological dose to plant operators or to the public will not be impacted as a result of the proposed change. Adequate SDM will continue to be ensured for all operational conditions.
>
> <u>DPC-NE-2011-P TS Changes</u>
> The proposed change revises the RNP and HNP TS to allow operation of a reactor core designed using the DPC-NE-2011-P methodology. As a portion of the overall Duke Energy methodology for cycle reload safety analyses, DPC-NE-2011-P has already been approved by the NRC for use at RNP and HNP. The proposed change will continue to ensure that applicable design and safety limits are satisfied such that the fission product barriers will continue to perform their design functions. Operation of the reactor in accordance with the DPC-NE-2011-P methodology will ensure the margin of safety is not reduced.
>
> Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

<u>Attorney for licensee</u>: Kathryn B. Nolan, Deputy General Counsel, Duke Energy Corporation, 550 South Tryon Street, Mail Code DEC45A, Charlotte NC 28202.

<u>NRC Branch Chief</u>: Undine Shoop.

<u>Duke Energy Progress, LLC, Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina</u>

<u>Date of amendment request</u>: October 10, 2017. A publicly-available version is in ADAMS under Accession No. ML17283A159.

Description of amendment request: The amendment would revise the Shearon Harris Nuclear Power Plant (HNP), Unit 1, Technical Specifications (TSs) to align more closely to improved Standard Technical Specifications for rod control and to the initial conditions in the HNP safety analyses. The proposed changes will delete TS action statement requirements that include a plant shutdown to address rods that are immovable but trippable. Revisions to surveillance requirements (SRs) are proposed to clarify actions that are not necessary if rods are immovable but still trippable.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed activity will delete action statement 3.1.3.1.c from the HNP TS and amend action statement 3.1.3.1.d, SR 4.1.1.1.1.a, and SR 4.1.1.2.a. These TS actions address electrical problems that prevent the Control Rod Drive Mechanism (CRDM) from moving rods. These conditions do not affect the safety functions of the control rods or shutdown margin of the unit. Rods will still insert into the core on an interruption of power to the CRDM, as occurs in a reactor trip. Also, rod alignment is not impacted, ensuring no change to reactivity.

   The proposed activity is removing actions from the HNP TS for conditions that do not impact the plant's safety analysis. Rods will still insert into the core on an interruption of power to the CRDM, as occurs in a reactor trip. Also, rod alignment is not impacted, ensuring no change to reactivity or shutdown margin. Since the conditions of these TS actions do not impact the plant safety analysis, the plant shutdown directed by them is unnecessary. The overall probability or consequence of an accident will not be significantly increased by removing the unnecessary TS actions.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed activity will delete action statement 3.1.3.1.c from the HNP TS and amend action statements 3.1.3.1.d, SR 4.1.1.1.1.a, and SR 4.1.1.2.a. These TS actions address electrical problems that prevent the CRDM from moving rods. These conditions do not affect the safety functions of the control rods. Rods will still insert into the core on an interruption of power to the CRDM, as occurs in a reactor trip. Also, rod alignment is not impacted, ensuring no change to reactivity or shutdown margin.

   The proposed change does not involve installation of new equipment or modification of existing equipment, so that no new equipment failure modes are introduced. Also, the proposed change in TS does not result in a change to the way that the equipment or facility is operated that would create new accident initiators.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed license amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed activity will delete action statement 3.1.3.1.c from the HNP TS and amend action statement 3.1.3.1.d, SR 4.1.1.1.1.a, and SR 4.1.1.2.a. These actions address electrical problems that prevent the CRDM from moving rods. These conditions do not affect the safety functions of the control rods. Rods will still insert into the core on an interruption of power to the CRDM, as occurs in a reactor trip. Also, rod alignment is not impacted, ensuring no change to reactivity or shutdown margin.

   The TS action statements as amended will continue to address the two required safety functions of rod control: to shut down the reactor in the event of a reactor trip, or to maintain proper

> alignment to ensure even power distribution. TS action statement 3.1.3.1.a will remain to drive actions if untrippable rods are identified. TS action statements 3.1.3.1.b and 3.1.3.1.d will remain to drive actions if misaligned rods are identified. The proposed changes to HNP TS do not significantly impact either rod safety function, and separate TS action statements for both functions will remain in place. Further, the impacted surveillances will continue to be applicable to conditions impacting either rod safety function.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Kathryn B. Nolan, Deputy General Counsel, Duke Energy Corporation, 550 South Tryon St., M/C DEC45A, Charlotte, NC 28202.

NRC Branch Chief: Undine Shoop.

Exelon Generation Company, LLC, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: October 31, 2017. A publicly available version is in ADAMS under Accession No. ML17304A984.

Description of amendment request: The amendment would revise Technical Specification (TS) Surveillance Requirement 3.8.4.3, "DC [Direct Current] Sources - MODES 1, 2, 3, and 4," for the R. E. Ginna Nuclear Power Plant (Ginna). The proposed change would allow the use of a consistent battery testing technique in order to provide consistent data for trending battery performance. This proposed change is based on guidance provided in the Institute of Electrical and Electronics Engineers (IEEE) Standard 450-2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," which is endorsed by NRC Regulatory Guide 1.129, Revision 3, "Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants."

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change will continue to ensure that the DC system is tested in a manner that will verify operability. Performance of the required system surveillances, in conjunction with the applicable operational and design requirements for the DC system, provide assurance that the system will be capable of performing the required design functions for accident mitigation and also that the system will perform in accordance with the functional requirements for the system as described in the Updated Final Safety Analysis Report for Ginna. This change is in accordance with IEEE Standard 450-2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," which has been endorsed by NRC Regulatory Guide 1.129, Revision 3, "Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants." This endures that the rate of occurrence and consequences of analyzed accidents will not change.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed surveillance requirement change will continue to ensure that the DC system and in particular the batteries are tested in a manner that will verify operability. No physical changes to the Ginna systems, structures, or components are being implemented. There are no new or different accident initiators or sequences being created by the proposed TS change. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change does not involve a significant reduction in the margin of safety. The proposed DC system surveillance requirement change provides appropriate and applicable surveillances for the DC system. The proposed change to surveillance requirements for the DC system will continue to ensure system operability.

   Therefore, this change does not affect any margin of safety for Ginna.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

<u>Attorney for licensee</u>: Tamra Domeyer, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

<u>NRC Branch Chief</u>: James G. Danna.

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units Nos. 1 and 2, Berrien County, Michigan

Date of amendment request: November 7, 2017. A publicly-available version is in ADAMS under Accession No. ML17317A472.

Description of amendment request: The proposed change would allow for deviation from National Fire Protection Association (NFPA) 805 requirements to allow for currently installed non-plenum listed cables routed above suspended ceilings and to allow for the use of thin wall electrical metallic tubing (EMT) and embedded/buried plastic conduit.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The use of EMT and embedded/buried PVC [polyvinyl chloride] does not create ignition sources and does not impact fire prevention. The EMT and embedded PVC had been in use since original plant construction, are allowed by the National Electrical Code and are not expected to increase the potential for a fire to start.

   The prior introduction of non-listed communication/data cables routed above suspended ceilings does not create ignition sources and does not impact fire prevention. Cable installation procedures are utilized to prevent the future installation of new cables that are noncompliant. Also, the communication/data cables routed above suspended ceilings do not result in compromising automatic fire suppression functions, manual fire suppression functions, fire protection or systems and structures, or post-fire safe shutdown capability.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes do allow future physical changes to the facility that deviate from NFPA 805 requirements. However, the proposed changes do not alter any assumptions made in the safety analyses, nor do they involve any changes to plant procedures for ensuring that the plant is operated within analyzed limits. As such, no new failure modes or mechanisms that could cause a new or different kind of accident from any previously evaluated are being introduced.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes do not alter the manner in which safety limits or limiting safety system settings are determined. No changes to instrument/system actuation setpoints are involved. The safety analysis acceptance criteria are not affected by this change and the proposed changes will not permit plant operation in a configuration outside the design basis.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Robert B. Haemer, Senior Nuclear Counsel, One Cook Place, Bridgman, MI 49106.

NRC Branch Chief: David J. Wrona.

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant (CNP), Units Nos. 1 and 2, Berrien County, Michigan

Date of amendment request: November 7, 2017. A publicly-available version is in ADAMS under Package Accession No. ML17317A454.

Description of amendment request: The proposed change would revise the CNP Emergency Plan to relocate the Technical Support Center (TSC) within the CNP protected area.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change to the CNP emergency plan to relocate the TSC does not impact the physical function of plant structures, systems, or components (SSC) or the manner in which SSCs perform their design function. The proposed change neither adversely affects accident initiators or precursors, nor alters design assumptions. The proposed change does not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an initiating event within assumed acceptance limits. No operating procedures or administrative controls that function to prevent or mitigate accidents are affected by the proposed changes.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not impact the accident analysis. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed or removed) or a change in the method of plant operation. The proposed change will not introduce failure modes that could result in a new accident, and the change does not alter assumptions made in the safety analysis. The proposed change to the location of the TSC is not an initiator of any accidents.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed change does not impact operation of the plant or its response to transients or accidents. The change does not affect the Technical Specifications or the operating license other than to amend the license to approve the change. The proposed change does not involve a change in the method of plant operation, and no accident analyses will be affected by the proposed changes.

   Additionally, the proposed change will not relax any criteria used to establish safety limits and will not relax any safety system settings. The safety analysis acceptance criteria are not affected by these changes. The proposed change will not result in plant operation in a configuration outside the design basis. The proposed change does not adversely affect systems that respond to safely shut down the plant and to maintain the plant in a safe shutdown condition. The emergency plan will continue to activate an emergency response commensurate with the extend of degradation of plant safety.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Robert B. Haemer, Senior Nuclear Counsel, One Cook Place, Bridgman, MI 49106.

NRC Branch Chief: David J. Wrona.

Southern Nuclear Operating Company, Docket Nos. 52-025 and 52-026, Vogtle Electric Generating Plant (VEGP), Units 3 and 4, Burke County, Georgia

Date of amendment request: October 6, 2017. A publicly-available version is in ADAMS under Accession No. ML17279B017.

Description of amendment request: The requested amendment proposes changes to the licensing basis documents to change the methodology and acceptance criteria for the in-containment refueling water storage tank (IRWST) heatup preoperational test described in the Updated Final Safety Analysis Report (UFSAR) Subsection 14.2.9.1.3, item h, and the passive residual heat removal (PRHR) heat exchanger preoperational test described in UFSAR Subsection 14.2.9.1.3, item g. These changes involve material which is specifically referenced in Section 2.D.(2) of the combined licenses for VEGP, Units 3 and 4.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   This activity changes the acceptance criteria for the IRWST heatup preoperational test and provides allowance to perform the preoperational test during both PRHR heat exchanger natural circulation and forced flow, instead of only during natural circulation. In addition, the acceptance criteria are changed for the PRHR heat exchanger forced flow system operability and preoperational tests.

   No structure, system, or component (SSC) or function is changed by this proposed activity. There is no change to the application of Regulatory Guide 1.68, nor is there a change to the design of the PRHR heat exchanger or the IRWST. The initial test program continues to confirm the heat transfer capability of the PRHR heat exchanger and that the IRWST heatup is consistent with the PRHR heat exchanger heat transfer modeling in the UFSAR Chapter 15 safety analysis.

   The proposed amendment does not affect the prevention or mitigation of abnormal events; e.g., accidents, anticipated operation occurrences, earthquakes, floods, turbine missiles, and fires or their safety or design analyses. This change does not involve containment of radioactive isotopes or have any adverse effect on a fission product barrier. There is no impact on previously evaluated accidents.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes do not involve a new failure mechanism or malfunction, that affects an SSC accident initiator, or interface with any SSC accident initiator or initiating sequence of events considered in the design and licensing bases. There is no adverse effect on radioisotope barriers or the release of radioactive materials. The proposed amendment does not adversely affect any accident, including the possibility of creating a new or different kind of accident from any accident previously evaluated. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

This activity changes the acceptance criteria for the IRWST heatup preoperational test and gives allowance to perform the preoperational test during both PRHR heat exchanger natural circulation and forced flow, instead of only during natural circulation. In addition, the acceptance criteria are changed for the PRHR heat exchanger forced flow system operability and preoperational tests.

No SSC or function is changed within this activity. There is no change to the application of Regulatory Guide 1.68, nor is there a change to how the PRHR heat exchanger or the IRWST are designed. The initial test program continues to confirm the heat transfer capability of the PRHR heat exchanger. The initial test program will confirm the IRWST heatup is consistent with the current PRHR heat exchanger heat transfer modeling in the UFSAR Chapter 15 safety analysis.

The proposed changes would not affect any safety-related design code, function, design analysis, safety analysis input or result, or existing design/safety margin. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the requested changes.

Therefore, the requested amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M. Stanford Blanton, Balch & Bingham LLP, 1710 Sixth Avenue North, Birmingham, AL 35203-2015.

NRC Branch Chief: Jennifer Dixon-Herrity.

Southern Nuclear Operating Company, Docket Nos. 52-025 and 52-026, Vogtle Electric Generating Plant, Units 3 and 4, Burke County, Georgia

Date of amendment request: November 16, 2017. A publicly-available version is in ADAMS under Accession No. ML17325A562.

Description of amendment request: The amendments propose changes to Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) in Combined License (COL) Appendix C, with corresponding changes to the associated plant-specific Tier 1 information to simplify and consolidate a number of ITAAC to improve efficiency of the ITAAC completion and closure process. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is also requested for the plant-specific Design Control Document Tier 1 material departures.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed non-technical change to COL Appendix C will consolidate ITAAC in order to improve and create a more efficient process for the ITAAC Closure Notification submittals. No structure, system, or component (SSC) design or function is affected. No design or safety analysis is affected. The proposed changes do not affect any accident initiating event or component failure, thus the probabilities of the accidents previously evaluated are not affected. No function used to mitigate a radioactive material release and no radioactive material release source term is involved, thus the radiological releases in the accident analyses are not affected.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change to COL Appendix C does not affect the design or function of any SSC, but will consolidate ITAAC in order to improve efficiency of the ITAAC completion and closure process. The proposed changes would not introduce a new failure mode, fault or sequence of events that could result in a radioactive material release.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change to COL Appendix C to consolidate ITAAC in order to improve efficiency of the ITAAC completion and closure process is considered non-technical and would not affect any design parameter, function or analysis. There would be no change to an existing design basis, design function, regulatory criterion, or analysis. No safety analysis or design basis acceptance limit/criterion is involved.

> Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: M. Stanford Blanton, Balch & Bingham LLP, 1710 Sixth Avenue North, Birmingham, AL 35203-2015.

NRC Branch Chief: Jennifer Dixon-Herrity.

Tennessee Valley Authority, Docket No. 50-391, Watts Bar Nuclear Plant, Unit 2, Rhea County, Tennessee

Date of amendment request: October 11, 2017. A publicly-available version is in ADAMS under Accession No. ML17284A452.

Description of amendment request: The amendment would revise Technical Specification (TS) 3.3.1, Table 3.3.1-1, "Reactor Trip System (RTS) Instrumentation," to increase the values for the nominal trip setpoint and the allowable value for Function 14.a, "Turbine Trip − Low Fluid Oil Pressure."

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

> 1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?
>
>   Response: No.

The proposed change reflects a design change to the turbine control system that results in the use of an increased control oil pressure system, necessitating a change to the value at which a low fluid oil pressure initiates a reactor trip on turbine trip. The low fluid oil pressure is an input to the reactor trip instrumentation in response to a turbine trip event. The value at which the low fluid oil initiates a reactor trip is not an accident initiator. A change in the nominal control oil pressure does not introduce any mechanisms that would increase the probability of an accident previously analyzed. The reactor trip on turbine trip function is initiated by the same protective signal as used for the existing auto stop low fluid oil system trip signal. There is no change in form or function of this signal and the probability or consequences of previously analyzed accidents are not impacted.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the [proposed] change create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No.

   The EHC [electrohydraulic control] fluid oil pressure rapidly decreases in response to a turbine trip signal. The value at which the low fluid oil pressure switches initiates a reactor trip is not an accident initiator. The proposed TS change reflects the higher pressure that will be sensed after the pressure switches are relocated from the auto stop low fluid oil system to the EHC high pressure header. Failure of the new switches would not result in a different outcome than is considered in the current design basis. Further, the change does not alter assumptions made in the safety analysis but ensures that the instruments perform as assumed in the accident analysis.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the [proposed] change involve a significant reduction in a margin of safety?

   Response: No.

   The change involves a parameter that initiates an anticipatory reactor trip following a turbine trip. The safety analyses do not credit this anticipatory trip for reactor core protection. The original

> pressure switch configuration and the new pressure switch configuration both generate the same reactor trip signal. The difference is that the initiation of the trip will now be adjusted to a different system of higher pressure. This system function of sensing and transmitting a reactor trip signal on turbine trip remains the same. There is no impact to safety analysis acceptance criteria as described in the plant licensing basis because no change is made to the accident analysis assumptions.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Sherry A. Quirk, Executive Vice President and General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, 6A West Tower, Knoxville, TN 37902.

NRC Branch Chief: Undine Shoop.

III. Notice of Issuance of Amendments to Facility Operating Licenses and Combined Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR chapter I, which are set forth in the license amendment.

A notice of consideration of issuance of amendment to facility operating license or combined license, as applicable, proposed no significant hazards consideration determination, and opportunity for a hearing in connection with these actions, was published in the *Federal Register* as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.22(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items can be accessed as described in the "Obtaining Information and Submitting Comments" section of this document.

<u>Entergy Nuclear Operations, Inc., Docket Nos. 50-003, 50-247, and 50-286, Indian Point Nuclear Generating Unit Nos. 1, 2, and 3, Westchester County, New York</u>

<u>Date of amendment request</u>: April 28, 2017, as supplemented by letters dated August 9, 2017; September 28, 2017; and October 26, 2017.

Brief description of amendments: The amendments revised the Cyber Security Plan Milestone 8 full implementation date by extending the full implementation date from December 31, 2017, to December 31, 2018.

Date of issuance: December 8, 2017.

Effective date: As of the date of issuance, and shall be implemented by December 31, 2017.

Amendment Nos.: 60 (Unit No. 1), 286 (Unit No. 2), and 263 (Unit No. 3). A publicly-available version is in ADAMS under Accession No. ML17315A000; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Provisional Operating License No. DPR-5 and Facility Operating License Nos. DPR-26 and DPR-64: The amendments revised the Provisional Operating License for Unit No. 1 and the Facility Operating Licenses for Unit Nos. 2 and 3.

Date of initial notice in *Federal Register*: July 18, 2017 (82 FR 32880).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 8, 2017.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50-255, Palisades Nuclear Plant, Van Buren County, Michigan

Date of amendment request: March 30, 2017, as supplemented by letter dated October 17, 2017.

Brief description of amendment: This amendment revised the Cyber Security Plan (CSP) implementation schedule Milestone 8 date and paragraph 2.E in the renewed facility operating license from December 15, 2017, to March 31, 2019. Milestone 8 of the CSP implementation schedule concerns the full implementation of the CSP.

Date of issuance: December 15, 2017.

Effective date: As of the date of issuance and shall be implemented within 30 days.

Amendment No.: 264. A publicly-available version is in ADAMS under Accession No. ML17328B033; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-20: Amendment revised the Renewed Facility Operating License.

Date of initial notice in *Federal Register*: May 23, 2017 (82 FR 23623). The supplemental letter dated October 17, 2017, provided additional information that expanded the scope of the application as originally noticed and changed the NRC staff's original proposed no significant hazards consideration (NSHC) determination as published in the *Federal Register*. Accordingly, the NRC published a second proposed NSHC determination in the *Federal Register* on November 7, 2017 (82 FR 51650). This notice superseded the original notice in its entirety. It also provided an opportunity to request a hearing by January 8, 2018, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 2017.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50-293, Pilgrim Nuclear Power Station (Pilgrim), Plymouth County, Massachusetts

Date of amendment request: March 30, 2017.

Brief description of amendment: The amendment revised Pilgrim's renewed facility operating license for the Cyber Security Plan (CSP) Milestone 8 full implementation completion date, as set forth in the CSP implementation schedule, and revised the physical protection license condition. The amendment revised the CSP Milestone 8 completion date from December 15, 2017, to December 31, 2020.

Date of issuance: December 15, 2017.

Effective date: As of the date of issuance, and shall be implemented within 30 days.

Amendment No.: 247. A publicly-available version is in ADAMS under Accession No. ML17290A487; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-35: The amendment revised the renewed facility operating license.

Date of initial notice in *Federal Register*: May 23, 2017 (82 FR 23624).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 2017.

No significant hazards consideration comments received: No.

National Institute of Standard and Technology (NIST), Docket No. 50-184, National Bureau of Standards Test Reactor (NBSR), Montgomery County, Maryland

Date of amendment request: March 2, 2017, as supplemented by letters dated March 29, 2017; May 25, 2017; November 17, 2017; November 20, 2017; December 1, 2017; December 11, 2017; and December 14, 2017.

Brief description of amendment: The amendment revised NIST NBSR's Facility Operating License TR-5 to allow receipt of calibration and testing sources, and revised technical specifications pertaining to the NIST reactor low power startup testing and organizational reporting requirements.

Date of issuance: December 15, 2017.

Effective date: As of the date of issuance.

Amendment No.: 11. A publicly-available version is in ADAMS under Accession No. ML17292A062; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Facility Operating License No. TR-5: Amendment revised the Renewed Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: September 12, 2017 (82 FR 42844). The supplemental letters dated November 17, 2017; November 20, 2017; December 1, 2017; December 11, 2017; and December 14, 2017 (which withdrew parts of the application), provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register.*

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 2017.

No significant hazards consideration comments received: No.

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit 1 (FCS), Washington County, Nebraska

Date of amendment request: December 16, 2016, as supplemented by letter dated May 15, 2017.

Brief description of amendment: The amendment revised the FCS Emergency Plan and Emergency Action Level (EAL) scheme for the permanently defueled condition. The proposed permanently defueled Emergency Plan and EAL scheme are commensurate with the significantly reduced spectrum of credible accidents that can occur in the permanently defueled condition and are necessary to properly reflect the conditions of the facility while continuing to preserve the effectiveness of the emergency plan.

Date of issuance: December 12, 2017.

Effective date: The amendment is effective April 7, 2018, and shall be implemented within 90 days of the effective date.

Amendment No.: 295. A publicly-available version is in ADAMS under Accession No. ML17276B286; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. DPR-40: The amendment revised the Emergency Plan and EAL scheme.

Date of initial notice in *Federal Register*: March 28, 2017 (82 FR 15383). The supplemental letter dated May 15, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 2017.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: March 27, 2017.

Brief description of amendment: The licensee requested to adopt NRC-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-535, Revision 0, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs" (ADAMS Accession No. ML112200436), dated August 8, 2011. The definition of shutdown margin in the Hope Creek Generating Station Technical Specifications is revised to require calculation of shutdown margin at the reactor moderator temperature corresponding to the most reactive state throughout the operating cycle, which is 68 degrees Fahrenheit or higher.

Date of issuance: December 13, 2017.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No.: 208. A publicly-available version is in ADAMS under Accession No. ML17317A605; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF-57: Amendment revised the Renewed Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: May 9, 2017 (82 FR 21560).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 13, 2017.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: March 27, 2017, as supplemented by letters dated April 28, 2017, and September 5, 2017.

Brief description of amendment: The amendment changed the Hope Creek Generating Station Technical Specifications (TSs) to relocate the reactor coolant system pressure-temperature (P-T) limit curves from the TSs to a new licensee-controlled document called the Pressure and Temperature Limits Report. The amendment also revised the 32 effective full power years P-T limit curves and approved P-T limit curves applicable through the license renewal term. The revisions to the curves were required due to the results of a recently pulled and tested reactor pressure vessel surveillance capsule.

Date of issuance: December 14, 2017.

Effective date: As of the date of issuance and shall be implemented within 60 days of issuance.

Amendment No.: 209. A publicly-available version is in ADAMS under Accession No. ML17324A840; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Renewed Facility Operating License No. NPF-57: Amendment revised the Renewed Facility Operating License and TSs.

Date of initial notice in *Federal Register*: May 23, 2017 (82 FR 23628). The supplemental letter dated September 5, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 14, 2017.

No significant hazards consideration comments received: No.

Southern California Edison Company, et al., Docket Nos. 50-206, 50-361, and 50-362, San Onofre Nuclear Generating Station, Units 1, 2, and 3, San Diego County, California

Date of amendment request: December 15, 2016, as supplemented by letter dated May 5, 2017.

Brief description of amendments: The amendments replaced the San Onofre Nuclear Generating Station, Units 1, 2, and 3 (SONGS) Permanently Defueled Emergency Plan and associated Emergency Action Level (EAL) Bases Manual (hereafter referred to as the EAL scheme) with an Independent Spent Fuel Storage Installation (ISFSI) Only Emergency Plan (IOEP) and associated EAL scheme. The NRC staff determined that the proposed SONGS IOEP and associated EAL changes continue to meet the standards in 10 CFR 50.47, "Emergency plans," and the requirements in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," of 10 CFR Part 50, as exempted. As such, the SONGS IOEP and associated EAL changes provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. These changes more fully reflect the status of the facility, as well as the reduced scope of potential radiological accidents once all spent fuel has been moved to dry cask storage within the onsite ISFSI, an activity which is currently scheduled for completion in 2019.

Date of issuance: November 30, 2017.

Effective date: As of the date Southern California Edison submits a written notification to the NRC that all spent nuclear fuel assemblies have been transferred out of the SONGS spent fuel pools and placed in storage within the onsite ISFSI, and shall be implemented within 60 days.

Amendment Nos.: 168 (Unit 1), 236 (Unit 2), and 229 (Unit 3). A publicly-available version is in ADAMS under Accession No. ML17310B482; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Operating License Nos. DPR-13, NPF-10, and NPF-15: The amendments revised the Facility Operating Licenses.

Date of initial notice in *Federal Register*: February 14, 2017 (82 FR 10601).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 30, 2017.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Docket Nos. 52-025 and 52-026, Vogtle Electric Generating Plant (VEGP), Units 3 and 4, Burke County, Georgia

Date of amendment request: May 10, 2017, and supplemented by letter dated September 20, 2017.

Description of amendments: The amendments consisted of changes to the VEGP, Units 3 and 4, Updated Final Safety Analysis Report in the form of departures from the incorporated plant-specific Design Control Document Tier 2* and Tier 2 information (text, tables, and figures). Specifically, the amendments consisted of changes related to revising the design reinforcement in the roof of the auxiliary building and the design of the girders supporting the roof.

Date of issuance: December 5, 2017.

Effective date: As of the date of issuance and shall be implemented within 30 days of issuance.

Amendment Nos.: 101 (Unit 3) and 100 (Unit 4). A publicly-available version is in ADAMS under Package Accession No. ML17311B236; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Combined Licenses No. NPF-91 and NPF-92: Amendments revised the Facility Combined Licenses.

Date of initial notice in *Federal Register*: June 6, 2017 (82 FR 26137). The supplemental letter dated September 20, 2017, provided additional information that clarified the application, did not expand the scope of the application request as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission’s related evaluation of the amendments is contained in a Safety Evaluation dated December 5, 2017.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Docket Nos. 52-025 and 52-026, Vogtle Electric Generating Plant (VEGP), Units 3 and 4, Burke County, Georgia

Date of amendment request: June 23, 2017.

Description of amendments: The amendments consisted of changes to the VEGP, Units 3 and 4, Updated Final Safety Analysis Report (UFSAR) in the form of departures from the plant-specific Design Control Document Tier 2 information and involves changes to the VEGP, Units 3 and 4, Combined License Appendix A, Technical Specifications (TSs). Specifically, the proposed changes revise plant-specific Tier 2 information to add the time delay assumed in the safety analysis for the reactor trip on a safeguards actuation (“S”) signal to UFSAR Table 15.0-4a. This is also reflected in the proposed revision to TS 3.3.4, “Reactor Trip System (RTS) Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” to add a surveillance requirement to verify the RTS response time for this “S” signal. The request also includes proposed changes to TS 3.3.7, “RTS Trip Actuation Devices,” to clarify that the requirements for reactor trip breaker (RTB) undervoltage and shunt trip mechanisms apply only to in-service RTBs. In addition, the request includes proposed changes to TS 3.3.9, “ESFAS Manual Initiation,” to correct the nomenclature for the Chemical and Volume Control System, which is inadvertently stated as the Chemical Volume and Control System.

Date of issuance: December 8, 2017.

Effective date: As of the date of issuance and shall be implemented within 30 days of issuance.

Amendment Nos.: 102 (Unit 3) and 101 (Unit 4). A publicly-available version is in ADAMS under Accession No. ML17296A236; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Combined Licenses No. NPF-91 and NPF-92: Amendments revised the Facility Combined Licenses.

Date of initial notice in *Federal Register*: August 15, 2017 (82 FR 38714).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 8, 2017.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Docket Nos. 52-025 and 52-026, Vogtle Electric Generating Plant (VEGP), Units 3 and 4, Burke County, Georgia

Date of amendment request: October 20, 2016.

Description of amendments: The amendments authorized changes to the Tier 2* information in the VEGP, Units 3 and 4, Updated Final Safety Analysis Report (which includes the plant-specific design control document information) to clarify the demonstration of the quality and strength of a specific set of couplers welded to carbon steel embedment plates, already installed and embedded in concrete through visual examination and static tension testing, in lieu of the nondestructive examination requirements of American Institute of Steel Construction (AISC) N690.

Date of issuance: September 5, 2017.

Effective date: As of the date of issuance and shall be implemented within 30 days of issuance.

Amendment Nos.: 86 (Unit 3) and 85 (Unit 4). A publicly-available version is in ADAMS under Package Accession No. ML17178A197; documents related to these amendments are listed in the Safety Evaluation enclosed with the amendments.

Facility Combined Licenses Nos. NPF-91 and NPF-92: Amendments revised the Facility Combined Licenses.

Date of initial notice in *Federal Register*: March 14, 2017 (82 FR 13662).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 5, 2017.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50-390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: March 31, 2017.

Brief description of amendment: The amendment revised Technical Specification (TS) 5.7.2.14, "Ventilation Filter Testing Program (VFTP)," to correct an administrative error introduced by Amendment No. 92, issued June 19, 2013. Specifically, Amendment 92 deleted TS 3.9.8, "Reactor Building Purge Air Cleanup Units," but did not delete associated references to the reactor building purge filters from TS 5.7.2.14.

Date of issuance: December 7, 2017.

Effective date: As of the date of issuance and shall be implemented within 30 days of issuance.

Amendment No.: 117. A publicly-available version is in ADAMS under Accession No. ML17311A786; documents related to this amendment are listed in the Safety Evaluation enclosed with the amendment.

Facility Operating License No. NPF-90: Amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in *Federal Register*: July 5, 2017 (82 FR 31103).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 7, 2017.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 21st day of December 2017.

For the Nuclear Regulatory Commission.

Kathryn M. Brock, Acting Director,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation.

[FR Doc. 2017-27930 Filed: 12/29/2017 8:45 am; Publication Date: 1/2/2018]